BNL-NUREG-69086

PERSPECTIVES ON SEISMIC CAPACITY OF US NUCLEAR PLANTS GAINED FROM THE IPEEE PROGRAM*

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ABSTRACT

As part of the U. S. Nuclear Regulatory Commission's (NRC's) program of Individual Plant Examination for External Events (IPEEE), all US licensees assessed their operating nuclear plants for severe accident risk due to seismic events and submitted the results to the NRC. Licensees used one of two methodologies to conduct their seismic IPEEEs. The first was a seismic probabilistic risk assessment consisting of at least a Level 1 analysis and a qualitative containment performance analysis. The second was a seismic margins assessment method, including a qualitative containment performance analysis. The scope of the seismic examination for a particular plant depended on the location of the plant, with higher hazard sites undertaking more extensive investigations. The NRC staff and its contractors reviewed all of the submittals to verify that the goals of the IPEEE program have been achieved, i.e., that the licensee's IPEEE process was capable of identifying seismic vulnerabilities as well as any cost-effective safety improvements. Perspectives and insights gained from this NRC review of the IPEEE seismic submittals are presented here.

KEYWORDS

Core damage, earthquake, high-confidence-low-probability-of-failure, nuclear, probabilistic, reactor, risk, seismic, seismic hazard, seismic margin

BACKGROUND

As part of the U. S. NRC's program of Individual Plant Examination for External Events (IPEEE), all US licensees assessed their operating nuclear plants for severe accident risk due to seismic events and submitted the results to the NRC. The NRC staff and its contractors reviewed all of these submittals to verify that the goals of the IPEEE program have been achieved, i.e., that the licensee's IPEEE process was

* This work was performed under the auspices of the U.S. Nuclear Regulatory Commission.

capable of identifying seismic vulnerabilities as well as any cost-effective safety improvements. Perspectives and insights gained from this review of the IPEEE seismic submittals are presented here.

As requested in NUREG-1407 (U.S. NRC, 1991), licensees used one of two methodologies to conduct their seismic IPEEEs. The first was a seismic probabilistic risk assessment (SPRA) consisting of at least a Level 1 analysis and a qualitative containment performance analysis. The second was a seismic margins assessment (SMA) method, including a qualitative containment performance analysis. The seismic margin could be characterized in terms of the high confidence of low probability of failure (HCLPF) capacity of each critical system, structure, and component (SSC) and the overall HCLPF of the plant. The HCLPF capacity can be defined as an earthquake acceleration level at which there is a 95% confidence that the given component, system, or plant being evaluated has a chance of failure of only 5%.

The principal products of an SPRA are an estimate of seismic core damage frequency (CDF), a list of dominant contributors to the seismic CDF, and a probabilistic plant-level capacity (i.e., fragility curve). For an SPRA, hazard estimates based on either Lawrence Livermore National Laboratory (LLNL) hazard curves (U.S. NRC, 1994), Electric Power Research Institute (EPRI) hazard curves (EPRI, 1989), or site-specific hazard curves were acceptable. The principal products of an SMA are a list of component capacities and an estimate of the HCLPF capacity of the plant. An SPRA and an SMA both satisfied the objectives of the seismic IPEEE, in that they both included a systematic, comprehensive walkdown of important components, and were both capable of identifying plant vulnerabilities.

The scope of the seismic examination for a particular plant depended on the location of the plant, with higher hazard sites undertaking more extensive investigations. Plants were grouped in NUREG-1407 into four categories: three levels of SMA (reduced scope, focused scope, full scope) or committed to perform an SPRA.

For reduced-scope plants, seismic capacities were verified to their existing design basis earthquake level, whereas focused scope and full-scope plants were evaluated against a review level earthquake (RLE) of 0.3g or 0.5g, depending on the plant's location and categorization in NUREG-1407. Full-scope and focused-scope plants had similar evaluation scopes, but the full-scope submittals involved more extensive evaluations of soil failures, relay chatter and component capacities. Licensees could, of course, conduct a more extensive evaluation than the minimum recommended level for their site. For example, an SPRA was acceptable for plants in all evaluation categories. Twenty-seven licensees, or about 40% of the total, conducted SPRAs, while the rest performed SMAs of varying scope: 4 full scope, 29 focused scope, and 12 reduced scope. (One licensee conducted both an SPRA and a focused scope SMA for its plant.)

PERSPECTIVES ON PLANT IMPROVEMENTS

Seventy percent of licensees identified a number of improvements to enhance the seismic ruggedness of their plants as a result of their seismic IPEEE analyses. The improvements fell into three categories: hardware modifications, improved procedures and training, and enhanced maintenance and housekeeping. The majority of the implemented or proposed improvements enhanced the plants' resistance to a seismic event, but did not involve significant cost.

Of plants that proposed plant improvements, 84% proposed some form of hardware changes, often adding new anchorages or supports, or strengthening existing ones. Many of these modifications were similar to ones carried out under the NRC's Unresolved Safety Issue (USI) A-46 program, but the IPEEE program involved additional SSCs, since the scope of the IPEEEs was broader than that of USI A-46.

About 60% of the plants that proposed plant improvements incorporated maintenance/housekeeping improvements, including the improvement of maintenance conduct and training; the correction of housekeeping errors; the issuance of new housekeeping standards; other corrective actions, such as restraining gas bottles, scaffolding, and ladders; corrective actions to address loose or missing fasteners, bolts, and clamps; and rust protection measures.

Finally, about 20% of plants that proposed plant improvements proposed revising or adding new procedures and training for seismic events.

In some cases, these plant improvements were only proposed in the submittals (sometimes without a firm commitment for implementation), while other submittals indicated the improvements were already implemented.

QUANTITATIVE RESULTS

CDF results from the SPRAs are shown in Figure 1. As indicated in this figure, most plants reported seismic CDFs between 1E-5 and 1E-4 per reactor-year (ry), with the next most common group falling between 1E-6 and 1E-5/ry. Only a few plants had CDFs higher than 1E-4/ry or less than 1E-6/ry. The use of the different hazard curves (EPRI vs. LLNL) did not affect the results significantly. The CDFs obtained from the SPRA analyses also indicated that the CDFs of newer plants (i.e., those designed and built to later seismic standards) were similar to the CDFs of older plants, built before some of the later design criteria were in place, but subjected to seismic backfit programs.

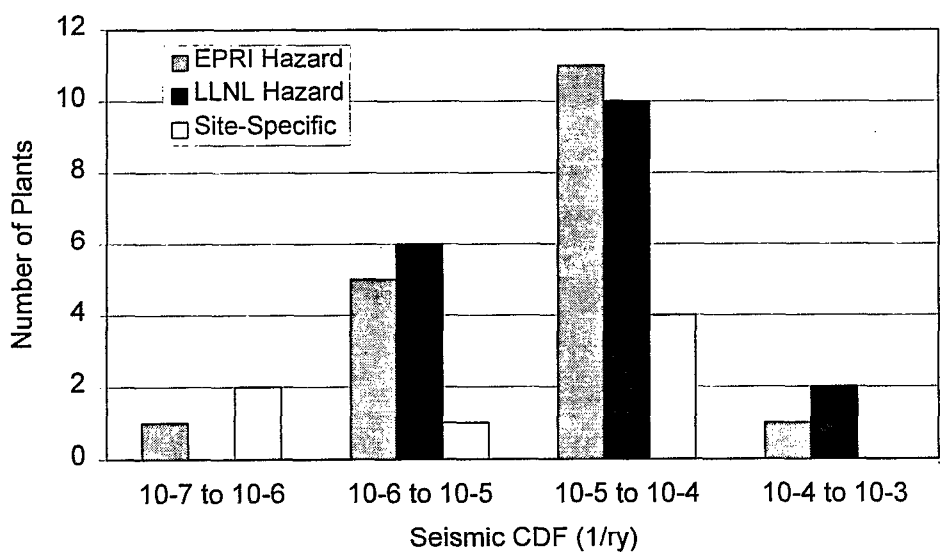

Figure 1: SPRA Results

Dominant contributors to seismic failure reported in the SPRAs involve the failure of the electrical systems, including the failure of offsite power; the failure of various electrical components, such as motor control centers, switchgears, relays, emergency diesel generators, and the dc batteries. Building and structural failures also contributed significantly.

Figure 2 shows the results from the seismic margin analyses in terms of the plant HCLPF capacities of the 36 plants that reported these capacities in their SMAs. Fourteen licensees reported plant HCLPFs of at least 0.3g, ten plants fell between 0.25 and 0.3g, nine plants were between 0.2 and 0.25g, and two plants were between 0.15 and 0.2g. One plant reported a HCLPF value of 0.12g. It should be noted that since a 0.3g RLE was used in the margin analyses, the calculations could not be used to demonstrate plant HCLPF capacity above 0.3g. As with the SPRA results, the seismic margins of older plants, built before some of the later design criteria were in place, are similar to the seismic margins of the newer plants.

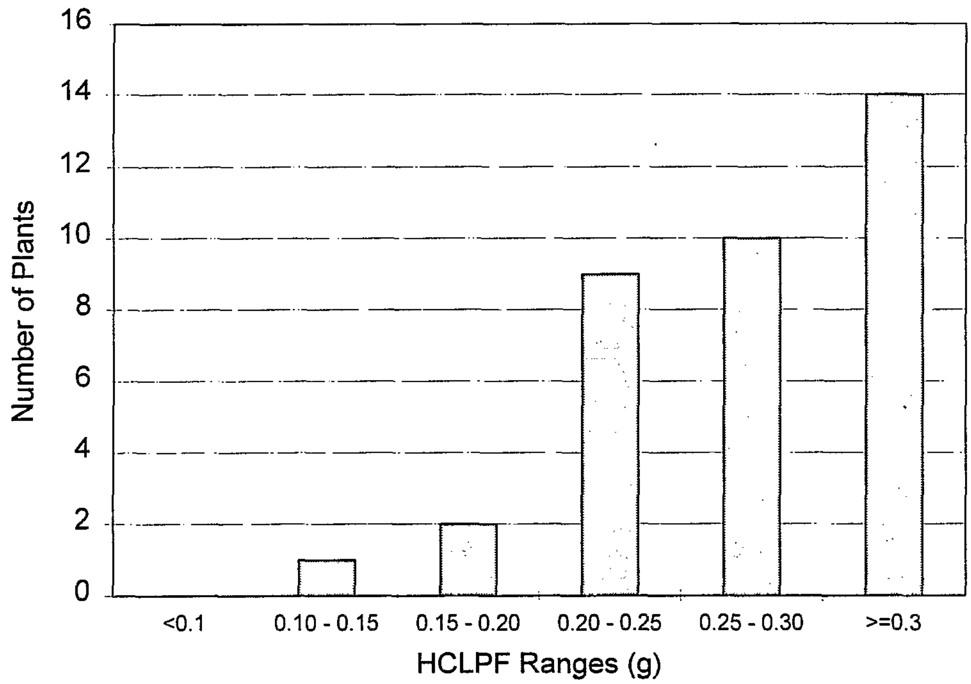

Figure 2: SMA Results

The HCLPF values reported in the SMAs generally fall into one of three categories: (1) a reported plant HCLPF at least equal to the 0.3g RLE, requiring some physical improvements of lower capacity components in most cases; (2) a plant HCLPF capacity less than the 0.3g RLE but greater than or equal to the safe shutdown earthquake (SSE) peak ground acceleration, requiring physical improvement of some low capacity components in some cases but not in others; and (3) a reported seismic capacity equal to the design basis SSE, mostly as a result of a reduced scope assessment.

To give an indication of the amount of seismic margin above the design level, the ratio of the plant HCLPF value to the plant SSE level is plotted in Figure 3. The figure also differentiates those plants that based their HCLPF calculations on a new structural analysis from those that based their HCLPF on scaling of the SSE results. The values shown in Figure 3 assume all proposed improvements are in place. The results indicate that the HCLPF value for all plants is never below the SSE and generally exceeds the SSE by a substantial margin.

Generally, the weak link or outlier components, i.e., those components governing the HCLPF capacity, identified in the SMA analyses were similar to the SSCs listed as dominant contributors to CDF in the SPRAs. The components identified as outliers in the SMAs included many electrical components and their anchorage, various tanks, residual heat removal (RHR) heat exchangers, and structures like the turbine and auxiliary buildings. Many licensees identified block walls located near safety-significant equipment as weak link structures.

Most licensees included seismic-fire and seismic-flood considerations within the scope of their analyses and in their seismic walkdown effort. In many of the analyses, the seismic-fire and/or seismic flood interaction evaluations revealed concerns and, in a number of instances, resulted in significant plant improvements.

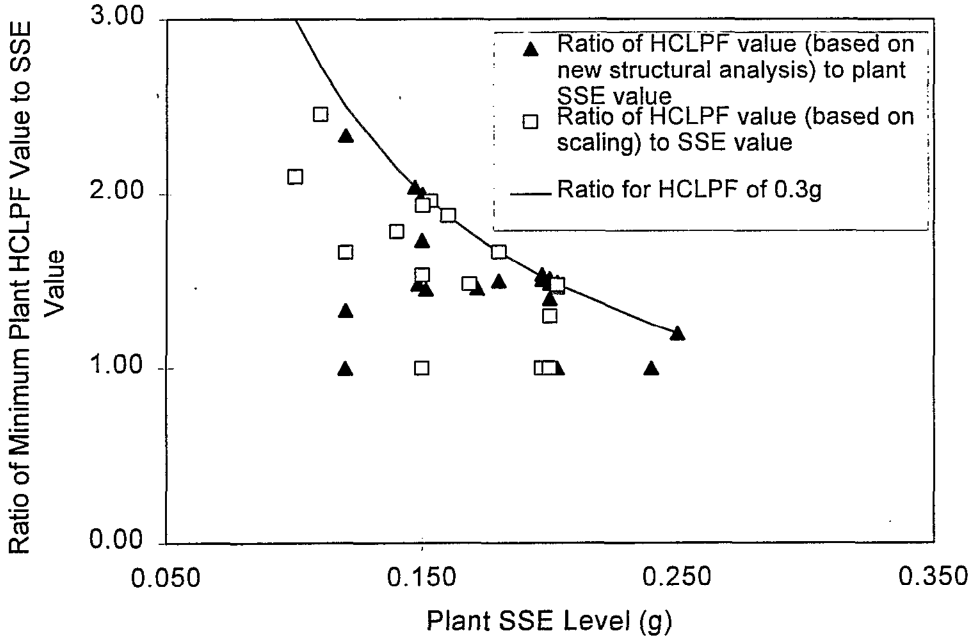

Figure 3: Comparison of Plant HCLPF level to SSE level

SEISMIC EVALUATION METHODS

The review of the IPEEE results, especially the SPRA results, indicated that the broad plant-to-plant variability in reported results was probably due to a combination of many factors, including differences in methods and analytical assumptions, as well as variations in plant designs and locations. In general, the assumptions and procedures were more consistent for HCLPF calculations in the SMAs than for the fragility calculations and other analyses in the SPRAs.

Some of the more important variations and observations identified in the reviews were as follows:

Licensees used a variety of hazard results to calculate CDFs. In many cases, the spectral shape used in evaluating fragilities, and the hazard curve used for quantifying CDF were not derived from a consistent set of hazard results. For example, the uniform hazard spectrum (UHS) derived from the 1989 LLNL hazard analysis (U.S. NRC, 1989) was typically used to define the SPRA spectral shape, whereas the seismic hazard curve derived from the 1989 EPRI or 1994 LLNL hazard analyses was typically used to quantify CDF.

Seismic fragility evaluations for the subset of components selected for risk quantification (components not screened) were carried out using various methods, including fragility analysis, generic information, and testing. In general, SPRAs liberally employed various forms of simplified fragility analyses, in contrast to the detailed conventional fragility analyses in past SPRAs. The use of simplified fragilities raises the possibility that in some cases the relative importance of the dominant contributors to the seismic CDF may be overstated or understated.

In calculating plant HCLPFs, the licensees made significant efforts to reduce inherent conservatism in the seismic demand calculations. Many of the plants performed new seismic analyses, and plant sites founded on soil usually considered soil-structure interaction (SSI) effects. As a result, many licensees demonstrated significant reductions in the seismic demands on plant components. Therefore, the HCLPF values computed are more realistic plant seismic capacities than conveyed by the design basis SSE capacities of the plants.

For the plants that performed new SSI analyses, the seismic analyses often resulted in much lower RLE in-structure response spectra (IRS) demand than the design basis SSE IRS. Therefore, comparisons of the component seismic fragility/HCLPF values for two plants using the two different approaches (scaling vs. new SSI calculations) could be misleading. The different approaches to estimating building and component seismic responses can significantly affect the magnitude of the reported fragility (or CDF) or HCLPF values. Hence, comparison of the seismic capacities should be made mainly among plants that were analyzed using similar methods.

The licensees of some plants in the eastern United States, when conducting their component fragility calculations, used UHS whose shapes differed from the conventional spectrum shapes derived from observed earthquakes. The energy content of these UHS appear to be reduced from that of the respective design basis SSE spectra, in the frequency range that is typically considered to have the greatest impact on the SSC responses to seismic motions. The seismic analyses using the UHS as input resulted in significant reduction in seismic demand, compared to the corresponding design basis calculations.

To date, an adequately detailed investigation of the implications of using one or more surrogate elements[1] in the IPEEE seismic analyses has not been undertaken. No regulatory guidelines have been developed concerning the use of surrogate elements (particularly with respect to sensitivities in plant logic modeling). If the surrogate element is found to be only a minor contributor to seismic CDF, then its use is probably reasonable. Since a surrogate element represents more than one component, those studies that identified the surrogate element as a dominant risk contributor could not determine which specific component among the surrogates is the risk contributor.

Finally, human actions were treated in SPRAs using a wide variety of approaches. In most SPRAs, human error probabilities were based on the values developed for the internal events models, usually modified with some simplified means for accounting for seismic-related performance shaping factors, and without strong technical bases for the values chosen. For SMAs, the timing and location of human actions was usually reported, along with qualitative comments on their reliability.

REFERENCES

EPRI (1989), "Probabilistic Seismic Hazard Evaluation at Nuclear Plant Sites in the Central and Eastern United States: Resolution of the Charleston Issue," EPRI NP-6395-D, April 1989.

U.S. NRC (1989), "Seismic Hazard Characterization of 69 Nuclear Power Plant Sites East of the Rocky Mountains," NUREG/CR-5250, LLNL, January 1989.

U.S. NRC (1991), "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, June 1991.

U.S. NRC (1994), "Revised Livermore Seismic Hazard Estimates of 69 Nuclear Plant Sites East of the Rocky Mountains," NUREG/CR-1488, LLNL, April 1994.

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[1] A surrogate element is an element used in a seismic PRA (SPRA) to account for the effects of the components that are screened out during the walkdown and screening phase of the SPRA. A surrogate element represents the failures of several screened components by the failure of a single surrogate element.